

ACCESSION #: 9010160047

LICENSEE EVENT REPORT (LER)

FACILITY NAME: Pilgrim Nuclear Power Station PAGE: 1 OF 13

DOCKET NUMBER: 05000293

TITLE: Manual Reactor Scram Due to Lockup of the Feedwater Regulating  
Valves

EVENT DATE: 09/02/90 LER #: 90-013-00 REPORT DATE: 10/02/90

OTHER FACILITIES INVOLVED: N/A DOCKET NO: 05000

OPERATING MODE: N POWER LEVEL: 060

THIS REPORT IS SUBMITTED PURSUANT TO THE REQUIREMENTS OF 10  
CFR SECTION:

50.73(a)(2)(iv)

LICENSEE CONTACT FOR THIS LER:

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Compliance Engineer

COMPONENT FAILURE DESCRIPTION:

CAUSE: X SYSTEM: BO COMPONENT: RLY MANUFACTURER: G080

E SJ FCV C635

E BN 12 T147

B BN INV A393

E SJ 90 M430

REPORTABLE NPRDS: Y

Y

Y

Y

Y

SUPPLEMENTAL REPORT EXPECTED: No

#### ABSTRACT:

On September 2, 1990 at 2233 hours, an unplanned manual reactor scram was initiated with reactor power at 60 percent. The operators manually scrammed the reactor due to difficulties experienced in controlling reactor vessel (RV) water level. A fuse blew in a feedwater control circuit power supply causing both Feedwater Regulating Valves (FRVs) to lockup without control room indication. A modification was implemented while shutdown which improves the reliability of the power supply and provides Control Room indication of a FRV lockup from a loss of control power.

After the shutdown other equipment problems were experienced. The startup FRV failed open due to air leaks and failure of its air booster relay that was later replaced. The Reactor Core Isolation Cooling System (RCICS) was declared inoperable due to the turbine tripping on three start attempts. The turbine trips were attributed to an improper manual start sequence specified in the RCICS operating procedure and/or looseness of the mechanical overspeed trip linkage. The RCICS operating procedure was revised and repairs were made to the mechanical overspeed trip linkage. The RCICS suction piping experienced a pressure transient due to the injection check valve not fully seating after the second start attempt. The check valve was modified to prevent recurrence. The High Pressure Coolant Injection System (HPCIS) was operable; however, the

turbine oversped and automatically reset on both starts. An exact cause of the overspeed trips could not be determined; however, corrective actions were taken for probable causes and the HPCIS was tested satisfactorily. A Residual Heat Removal System/Shutdown Cooling suction isolation valve would not open normally. The cause was contact failure on the valve's seal-in relay (General Electric type CR120A) that was replaced. This report is submitted in accordance with 10 CFR 50.73(a)(2)(iv) and this event posed no threat to the public health and safety.

END OF ABSTRACT

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Component Failure Description appended on LER form.

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#### EVENT DESCRIPTION

On September 2, 1990 at 2233 hours, an unplanned manual reactor scram was initiated while at 60 percent reactor power. The operators manually scrammed the reactor due to difficulties experienced in controlling Reactor Vessel (RV) water level. On September 2, 1990 at 2155 hours the Control Room received the "Reactor Water Hi/Lo Level" annunciator on panel C-905L. The operators observed a slow unexplained increase in the RV water level on level recorder 640-26. The high RV water level alarm setpoint is calibrated at inches. The master feedwater controller (set at 27 inches) was sending a close signal to the Feedwater Regulating Valves (FRVs). The operators took manual control of the Feedwater

Control System attempting to restore RV water level to normal. However, the RV water level continued to rise. The operators noted no increase in steam flow and verified the instrument air pressure was satisfactory. The operators also observed the FRV lockup lights, located above the FRV controllers, were not lit indicating no lockup signal was present. The operators attempted to reset the FRV lockup per procedure 2.4.49 (Rev. 15), "Loss of Normal Feed and Feedwater Control Valve Malfunction". The operators opened the "A" and "B" Reactor Feedpumps' (RFPs) minimum flow valves to reduce RV water level. Additionally, the operators attempted to take individual control on each FRV. Changes were observed in the FRVs' (FV-642A/B) position demand signal but the feedwater flow did not change. From 2158 hours to 2233 hours the operators cycled the RFP minimum flow valves to control RV water level while attempting to regain FRV control. When the operators determined that FRV control could not be re-established the "C" RFP was tripped, the speed of each recirculating pump was reduced to approximately 50 percent and the reactor mode selector switch was moved from the RUN position to the SHUTDOWN position.

As expected, the RV water level decreased in response to the scram due to shrink (i.e., decrease in the void fraction in the RV water). The RV water level decreased to approximately -10 inches. The low RV water level setpoint is calibrated at approximately inches; therefore, the level decrease resulted in automatic actuations of the Primary Containment Isolation Control System (PCIS) and Reactor Building Isolation Control System (RBIS).

The PCIS actuation resulted in the following designed responses:

- o Automatic closing of the inboard and outboard Primary Containment System (PCS) Group 2 (two) isolation valves that

were open.

- o The PCS Group 3 (three)/Residual Heat Removal System (RHRS) isolation valves, in the closed position, remained closed.

- o Automatic closing of the inboard and outboard PCS Group 6 (six)/Reactor Water Cleanup (RWCU) System isolation valves.

The RBIS actuation resulted in the automatic closing of the Reactor Building/Secondary Containment System (SCS) supply and exhaust ventilation dampers (trains 'A' and 'B'), and the automatic start of trains 'A' and 'B' of the SCS/Standby Gas Treatment System (SGTS).

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The operators followed procedure 2.1.6, "Reactor Scram", and EOP-1, "RPV Control". The RWCU System was restored to service and the "A" RFP was started. The RHRS loop "A" was placed in service in the Suppression Pool Cooling (SPC) mode in anticipation of starting the Reactor Core Isolation Cooling System (RCICS). The downstream feedwater block valves (MO-3479 and MO-3480) were closed and the Reactor Protection System (RPS) was reset at 2242 hours. The operators attempted to start the RCICS for RV water level control in accordance with procedure 2.2.22, [RCICS], Rev. 33. However, the turbine tripped on overspeed. The overspeed was reset and the operators attempted to start the RCICS for RV water level control for a second time and the turbine tripped. The RCICS was successfully started on the third attempt in the full flow test mode. However, the turbine tripped when the operators attempted to shift over for RV water level control. During this time, the operators attempted to control RV water level with the startup FRV (FV-643). However, the operators could

not control the startup FRV position. The RBIS was reset and the SGTS was secured. The RV water level decreased below inches at 2314 hours which again initiated the PCIS, RBIS and a RPS scram signal. The low water level initiated the designed responses as described earlier. The RPS was subsequently reset.

The operators proceeded to control RV water level by starting and stopping the RFPs. The RV water level increased to inches which initiated a PCIS Group 1 isolation signal. This resulted in the closing of the Main Steam Isolation valves (MSIVs) and an RPS scram signal because the MSIVs were closing (i.e., less than 90% open) with main steam line pressure greater than 600 psig. The operators cycled the Main Steam/Target Rock two-stage relief valves RV-203-3B (S/N 1048) and RV-203-3C (S/N 1046) to lower reactor pressure since the MSIVs were closed. The operators continued to cycle the RFPs for RV water level control. The operators entered EOP-03, "Primary Containment Control", due to Suppression Pool high temperature and placed the "B" RHR pump in service (both loops of RHRS were then in the SPC mode).

On September 3, 1990 at 0017 hours the RV water level decreased below inches again which actuated the PCIS Groups 2, 3, and 6 circuitry. The isolations functioned as designed. The High Pressure Coolant Injection System (HPCIS) was started for RV water level control and was run for approximately two minutes and then secured. The HPCIS was later started in the full flow test mode for pressure control at 0043 hours. The operators noted speed and flow oscillations with the controller in automatic at a speed of approximately 2600 rpm. The controller was taken to manual which eliminated the oscillations. The HPCIS ran for approximately 3 hours in the manual control mode before being secured. The operators then reset the RBIS and secured the SGTS. The PCIS was

reset and RWCU System was returned to service. The MSIVs were opened, the RPS was reset. Reactor pressure was lowered using the Main Steam/Turbine Bypass Valve Opening Jack (BVOJ).

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At 1115 hours, the operators attempted to place the "A" loop RHRS into the Shutdown Cooling (SDC) mode. However, the inboard SDC suction piping isolation valve (MO-1001-50) would not fully stroke open when the control switch was moved to the open position and released. The valve was subsequently opened at 1411 hours by holding the control switch in the open position. The "A" RHR pump was started at 1450 hours for shutdown cooling. However, a PCIS Group 3 isolation occurred and the inboard and outboard SDC suction isolation valves closed. This event is reported separately via LER 50-293/90-014. The "A" RHRS pump was successfully started for shutdown cooling at 1733 hours.

The NRC Operations Center was notified in accordance with 10 CFR 50.72 at 0102 hours on September 3, 1990. A Multidisciplinary Analysis Team (MDAT) was formed to investigate the event in accordance with procedure 1.3.37, "Post Trip Reviews".

## CAUSE AND CORRECTIVE ACTION

The individual causes and their corrective actions are discussed together for clarity and are as follows:

### 1. FRVs

The operators could not control the FRVs because the FRVs were locked up

due to a blown fuse in the feedwater lockup solenoids' power supply. During a routine plant walkdown on August 12, 1990, the "B" FRV (FV-642B) was identified as having a packing leak. A Maintenance Request was written to repair the leak during the next unit outage. In the interim, the leak was monitored daily. On August 29, 1990, the leak worsened to a point where repair was necessary. The packing leak was repaired on August 30 and 31, 1990 by injecting a sealing compound. The equipment and instrumentation around FV-642B were subjected to steam and moisture from the packing leak. After the leak repair, an inspection was conducted to identify any obvious equipment damage from the packing leak. However, moisture penetrated a junction box, located near FV-642A, due to a degraded seal. The moisture inside the junction box later migrated down a conduit and collected inside Pressure Switch PS-656A. The moisture inside PS-656A grounded the switch contacts which caused the power supply fuse to blow. This de-energized the feedwater lockup solenoids SV-655A and SV-655B which isolated the instrument air supply to PS-656A/B and air lock relays AL-642A/B. When the air lock relays actuated, the pressure inside the actuators of FRVs (FV-642A/B) became trapped, as designed, to prevent valve movement. However, FV-642A slowly drifted open due to air leakage (approximately 1 psi/minute) which resulted in the reactor water level slowly increasing. The operators were unaware the FRVs were locked up because the lockup indicating lights on the Main Control Board (C-905) were powered from the same power supply that de-energized because of the blown fuse.

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Investigation determined the packing leak on FV-642B was due to a degradation of the valve stem and packing box. Indications were found that the stem and the stuffing box were scored and gouged. The stuffing

box scoring was probably the result of water flashing during previous packing leaks. Leak repair sealant removal may have caused the stuffing box gouges. The degraded condition of the electrical junction box that feeds PS-656A and SV-655A was due to a lack of a detailed inspection after the FV-642B leak repair and a lack of restoration of the junction box seal during previous work. Although an inspection was conducted, it was not of sufficient detail to identify the problem. The Feedwater Control System is non-safety related and the original design did not provide separation of power for the FV-642A and FV-642B control circuits. This allowed a single failure (blown fuse internal to the power supply) to lockup both FRVs. Additionally, the original design did not include a FRV lockup alarm due to a loss of power. The operators could not control RV water level with the startup FRV (FV-643) because the valve locked up and failed open due to air leakage. The lockup of the startup FRV was due to the failure of its air booster relay. The startup FRV air system was noted as having an air leak on August 26, 1990. A Maintenance Request was initiated and planned to be worked during the next power reduction due to ALARA considerations. The leak worsened until the valve locked up. The valve eventually drifted open due to air leakage. The air booster relay failed due to internal diaphragm degradation. This may have been related to the age of the relay and its location in the condenser bay environment.

The FRV power supply (640-42) fuse was replaced. The pressure switch (PS-656A) was replaced. The FRVs (FV-642A/B) and the startup FRV (FV-643) were repacked and repairs made to minimize scoring on the FV-642B stuffing box. Air leaks were identified and repaired. The FV-642A air actuator was replaced. The air booster relay for FV-643 was replaced. The air booster relay was manufactured by the Moore Products Company (F/R Booster, Model 61H, B/M 10342S16BH). The degraded junction

box seal was replaced and other junction boxes in the area were inspected for indication of seal degradation. Any degraded seals identified were replaced. Additionally, a Request for Investigation (RFI 90-722) was initiated to consider development of guidelines addressing large leaks providing structured guidance for required actions and depth of inspections. Weep holes were drilled in the junction boxes for FV-642A and FV-642B to prevent water buildup. A modification (PDC 90-56) was implemented while shutdown. This modification separated the power supply output to the control circuitry for the FRVs. Prior to this modification, the FV-642A/B lockup control circuits were powered from the same power supply output. This modification provides improved system reliability in that a single failure of the power supply output will not cause a lockup of both FRVs. This modification also added Control Room indication for a FRV lockup due to a loss of power. The preventive maintenance program for the FRVs and junction boxes will be reviewed and revised as necessary.

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## 2. RCICS

The first trip of the RCICS turbine was due to an actual overspeed condition. Plant information computer (EPIC) traces showed turbine speed reaching 5725 rpm. The overspeed trip setpoint (5512 rpm to 5737 rpm) is approximately 125% 2% of rated speed (4500 rpm). The overspeed occurred as a result of the manual injection step sequence specified in procedure 2.2.22, "Reactor Core Isolation Cooling System", Rev. 33. Section 7.4 describes the process for manual operation to inject into the reactor vessel. The turbine steam inlet valve (MO-1301-61) is opened and the injection valve (MO-1301-49) is opened after pump discharge pressure

equals reactor pressure. This sequence was specified to avoid overpressurizing the RCICS suction line during manual starts if the injection check valve (1301-50) is not fully seated. The computer traces showed the turbine steam inlet valve was opened and the turbine reached overspeed before the injection valve was opened. When the steam inlet valve was opened all the pump flow was directed to the Suppression Pool via the minimum flow bypass line. The flow element (FE 1360-3) provides input to the RCICS turbine control logic via the flow transmitter (FT 1360-4) and is located downstream of the minimum flow bypass line. Therefore, the turbine governor valve received input to open further in an attempt to achieve rated flow and the turbine tripped on overspeed prior to the injection valve opening. A contributing cause was the verification/validation of procedure (2.2.22). The operating sequence for injection identified in the procedure was a corrective action due to the April 12, 1989 RCICS suction line pressure transient when the 1301-50 check valve failed to close. The procedure change was validated using the simulator. However, the modeling of this scenario indicated the simulator response was slower than that observed in the plant.

The second and third RCICS turbine trips were caused by looseness in the mechanical overspeed trip linkage. The looseness resulted from repeated closing of the trip and throttle valve, normal system vibration and a slight rounding of the tappet nut latching surface. Additionally, a substance was found in the tappet guide of the tappet and ball assembly. This caused sluggish reset action resulting in improper resetting. The substance is believed to be from a joint sealer used on the turbine inspection panels. The turbine speed in the second trip reached a maximum of 4390 rpm which is below rated speed (4500 rpm). The computer traces show the injection valve was opened soon enough to prevent the overspeed condition experienced on the first start attempt. RCICS flow

to the RV was established for a short period of time before the trip and throttle valve closed. The third trip occurred at a turbine speed of approximately 4000 rpm with the RCICS in the full flow test mode.

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Contributing causes to the overspeed include the method for RCICS turbine shutdown. The vendor representative (General Electric) recommended closing the trip and throttle valve only when required by the Inservice Testing (IST) program. This will significantly reduce mechanical overspeed trip linkage vibration. Another contributing cause was the failure of plant personnel to recognize deficient conditions with the trip linkage. A similar event occurred on August 23, 1990 during surveillance testing. The turbine tripped two times at less than overspeed conditions. The cause was then believed to be improper resetting of the mechanical overspeed trip linkage.

The RCICS procedure 2.2.22, was revised (to Rev. 34) to reduce the likelihood of an overspeed trip during a manual start for RV injection. Additionally, information on how to reset the trip and throttle valve was added. Similarly, changes were made to the HPCIS operating procedure, 2.2.21, "High Pressure Coolant Injection (HPCI) System", and included in Rev. 39, to reduce the likelihood of overspeed tripping.

The RCICS mechanical overspeed trip mechanism (weight and reset spring) were disassembled and cleaned. The overspeed trip linkage was cleaned, the tappet nut latching surface was machined, and the linkage was adjusted. The RCICS oil sumps and piping were cleaned, flushed and the oil was changed.

Surveillance procedure 8.5.5.1, "RCIC Pump Operability Flow Rate and Valve Test at Approximately 1000 PSIG", was revised (to Rev. 30) to change the method of securing the RCICS turbine reduce trip and throttle valve closures.

In addition to these contributing causes, the oil sampling and analysis program did not correct the presence of sludge in the RCICS oil in a timely manner. The oil sampling and analysis program will be reviewed and appropriate changes made to ensure timely identification of potential problems.

During RCICS walkdowns, the suction piping pressure indicator PI-1360-20 was found stuck at full scale (100 psig). Review of computer traces indicated a malfunction of the RCICS injection check valve 1301-50 (Anchor Darling Model E3112-3, four inch swing type) during the second RCICS turbine trip. This subjected the RCICS piping to a pressure transient. The suction piping was pressurized after the turbine tripped until the injection valve (MO-1301-49) was fully closed. The injection check valve did not close in time to prevent the suction piping from being pressurized. Field inspection of the as-found condition indicated the check valve was open approximately one and one-eighth inches. This valve is a testable check valve where the disc can be moved externally by turning the shaft. The most probable cause of the valve not fully seating was the shaft key, that allows exercising, was loose and tended to bind resulting in the disc and shaft moving together. This made the disc rotate against the packing load rather than freely on the shaft. Contributing to the problem was the bushing turning with the hinge pin which was a design feature, intended by the valve manufacturer, to facilitate valve position indication. This design feature is not used at Pilgrim Station.

The injection check valve (1301-50) was modified (FRN 90-03-84, 85, 86 and 87) to reduce friction allowing the disc to move freely. A walkdown of RCICS piping for structural integrity found the piping and supports intact. A magnetic particle exam was performed on a RCICS butt weld (1-3-D) since this was calculated as the highest stress weld for this transient. A linear surface indication, identified as a manufacturing defect, was discovered and removed. A piping analysis was completed for the suction and discharge piping. The analysis showed the piping was not stressed beyond the code allowable limits.

The RCICS was tested satisfactorily during plant startup to verify proper operation of the overspeed trip mechanism and linkage. Additionally, procedure TP 90-68 (Rev. 1) "RCIC Vessel Injection and Discharge Check Valve Alternate Safety Position Verification," was completed on September 25, 1990. This test started the RCICS for RV injection which provided a forward flow exercise for the injection check valve (1301-50). Following RCICS turbine shutdown, the check valve was verified to provide reverse flow protection. The check valve (1301-50) will be disassembled and inspected during the next scheduled outage (Refueling Outage No. 8).

### 3. HPCIS

During HPCIS RV water level control operation, flow oscillations were noted. Additionally, when HPCIS was placed in the full flow test mode for RV pressure control, flow oscillation was observed in the automatic control mode at a speed of about 2600 rpm (i.e., approximately 2800 gpm). Review of computer traces showed the flow oscillations during injection

were due to system perturbations caused by opening of the test line throttle valve (MO-2301-10). Review of the HPCIS vendor manual revealed the EG-R hydraulic actuator needle valve was not adjusted correctly, i.e. the valve was found one full turn open instead of one-quarter turn open. This valve was adjusted and smoother operation was noted during subsequent testing. Contributing to the flow oscillations was the limitation of the normal control system capability. The original HPCIS design was for full flow injection. The flow control system was not designed for low flow/rpm operation. Operating procedure 2.2.21, was revised (to Rev. 39) to limit automatic operation to 3000 gpm or greater conditions. Procedure 8.E.23.1. (Rev. 0), "HPCI Turbine Speed Control System Calibration", will be revised to include an as-found and as-left position for the EG-R needle valve. Engineering Service Request (ESR 90-387) was written to consider replacement of the HPCI flow controller with a current design to improve the turbine operating range and flow stability in the automatic control mode.

During both HPCIS starts, an overspeed trip occurred. The trip automatically reset and HPCIS automatically restarted successfully. The overspeed trip occurred at approximately 125 percent of rated speed, as designed. Root cause investigation for the HPCIS overspeed trips did not identify a specific root cause. However, a review of the turbine oil system valve lineup identified the oil relay pilot supply block valve (HO-2301-123) was open, as specified in procedure 2.2.21 (Rev. 38), "High Pressure Coolant Injection System". The vendor had previously determined the valve should be one-half to one turn open. The valve was repositioned to one-half turn open and procedure 2.2.21 (Rev. 39) was changed (SRO 90-218) to set the valve at one-half turn open.

Additionally, there are indications that there was sludge in the oil sump. Laboratory analysis showed the oil contained organic impurities, was acidic, and had significant water content. The system was flushed and the oil was changed.

Instrumentation was connected (TM 90-31) for HPCIS diagnostic testing. The testing was completed on September 23, 1990 which indicated smoother HPCIS operation. The HPCI surveillance schedule was revised to perform operability testing every two weeks for a period of two months. Two additional parameters will be monitored to further help identify the cause if another overspeed trip occurs. The additional instrumentation will be used to monitor test results for proper system operation. This LER will be updated if significant corrective actions are necessary as a result of the augmented testing.

#### 4. RHRS/SDC Inboard Suction Isolation Valve MO-1001-50

Valve MO-1001-50 did not stroke to the full open position when the operator used the control switch on the control room panel (C-903). This was indicated by both valve position lights being illuminated on panel C-903. The operators unsuccessfully attempted to open the valve several times. Using the normal opening method, the control switch is moved to the open position and released with the switch being returned to neutral by spring action. The valve was later fully opened by holding the control switch in the open position. The valve later closed automatically on a PCIS Group 3 isolation signal (see LER 50-293/90-14). The valve was subsequently reopened without holding the control switch in the open position. The failure of the seal-in circuit could not be duplicated during troubleshooting, but apparently cleared subsequent to

the Group 3 isolation and reset. Testing of the seal-in relay (16A-K29) contact showed fluctuating resistance values which indicated a possible contact failure. Therefore, the most probable cause of the problem with opening valve MO-1001-50 was the failure of contacts 5 and 6 of relay 16A-K29 (General Electric CR120A). This was determined to be an isolated failure. The valve's breaker (52-2046) open coil and auxiliary contactor (42-2046) was replaced, and the relay (16A-K29) was replaced and post work tested satisfactorily.

#### 5. Safety Valve RV-203-4B

A review of temperature recorder (260-20HG) indicated an increase in RV-203-4B tailpipe temperature from 165 degrees Fahrenheit to 185 degrees Fahrenheit, beginning approximately 30 seconds before the scram. The setpoint of RV-203-4B is 1240 psig 12 psig. Reactor pressure was 1036 psig at the time of the scram and decreased after the scram. A visual inspection of the valve was conducted on September 5, 1990 and no indications were observed that the valve opened or leaked. On September 22, 1990 during a drywell inspection with the RV pressurized, the RV 203-4B was inspected and no leakage was identified. However, valve HO-1301-90 was found to be leaking from the packing gland. This valve is on a vent line from the RCICS turbine steam supply piping, and is located near the RV-203-4B tailpipe. The steam leak was directed at the tailpipe, and therefore was determined as the cause of the tailpipe temperature increase. The packing gland for valve HO-1301-90 was tightened to stop the leak.

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Maintenance Program Review

An extensive review and self-assessment of the Maintenance Program and practices at Pilgrim Station is currently underway. Further Maintenance Program improvements may be identified and implemented as a result of this assessment.

## SAFETY CONSEQUENCES

This event posed no threat to the public health and safety.

The RV water level remained well above the top of active fuel zone at all times during this event. There were no radiological releases due to this event.

All of the Core Standby Cooling System (CSCS) subsystems were operable during this event. The CSCS consists of the HPCIS, Automatic Depressurization System (ADS), Core Spray System (CSS) and the RHRS/Low Pressure Coolant Injection (LPCI) mode. The HPCIS was manually initiated twice, once for RV water level control and once for RV pressure control. Problems were noted with the HPCIS turbine tripping on overspeed and flow oscillations. An evaluation was conducted which concluded the HPCIS was operable and capable of delivering rated flow (4250 gallons per minute) within the required time frame (90 seconds from initiation signal).

An analysis was done for the RCICS pressure transient. The analysis concluded the stresses experienced were below the code allowable stresses and therefore, the RCICS suction and discharge piping was not overpressurized.

The closing of the MSIVs was the designed response to the high RV water

level. The highest RV water level that occurred was approximately inches which is 34 inches below the main steam line nozzles.

The safety design basis for the RHRS/SDC inboard suction isolation valve (MO-1001-50) is to close within 30 seconds of a PCIS Group 3 (three) isolation signal. The inability to open the valve normally did not affect the valve's ability to meet the safety design basis.

The Suppression Pool temperature and water level increased due to HPCIS, RCICS, and relief valve(s) operation. The maximum Suppression Pool temperature that occurred while shutdown was approximately 96 degrees Fahrenheit which is less than the limit (120 degrees) specified by Technical Specification 3.7.A.1.h. The maximum Suppression Pool water level that occurred while shutdown was approximately 132 inches (LI-1001-604) which is less than the level (approximately 139 inches) corresponding to the maximum volume (94,000 cubic feet) specified by Technical Specification 3.7.A.1.b.

This report is submitted in accordance with 10 CFR 50.73(a)(2)(iv) due to a manual actuation of the Reactor Protection System (RPS), the automatic actuations of the RPS, PCIS and RBIS logic circuitry, and the manual initiation of HPCIS for RV water level control.

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#### SIMILARITY TO PREVIOUS EVENTS

A review was conducted of Pilgrim Station Licensee Event Reports (LERs) submitted since January 1984. The review focused on LERs submitted in accordance with 10 CFR 50.73(a)(2)(iv) that involved problems with

feedwater control or high RV water level. Additionally, LERs involving RCICS, HPCIS and RHRS/SDC were analyzed. The review identified events reported in LERs 50-293/85-008-01, 88-024-00, 89-014-01, 89-015-00 and 89-028-00.

For LER 85-008-01, the HPCIS turbine tripped on overspeed due to a faulty connector in the HPCIS turbine control system. The cable between the Electric Governor control assembly (EG-M) and the Electric Governor actuator assembly (EG-R) became disconnected at the EG-R. This was the result of a broken retaining ring in the female connector at the EG-R end of the cable. The faulty cable connector was replaced.

For LER 88-024-00, the MSIVs closed due to a high RV water level. The cause of the high water level was a pin that became disassociated from the feedback cam linkage of the valve positioner for valve FV-642A. The linkage for FV-642A was reconnected and secured via two locknuts. FV-642B was inspected with satisfactory results.

For LER 89-014-01, the RCICS suction piping was overpressurized on April 12, 1989 due to operator error. Additionally, the RCICS injection check valve (1301-50) failed to close. The valve did not close due to leak seal material in the valve's actuator shaft bushing. The check valve was opened and visually inspected. The valve internals affected by the leak seal material were replaced and the valve disc was reinstalled. Additionally, the RCICS piping was analyzed and code evaluated. The analysis concluded the pressure transient did not impair the integrity of the RCICS piping components and the suction piping did not need to be replaced. That analysis was used for comparison of the September 2, 1990 pressure transient. The evaluation concluded the September 2, 1990 pressure transient was much less severe than the April 12, 1989 event

since computer traces showed that this pressure transient was not instantaneous.

For LER 89-015-00, an automatic turbine trip, generator trip, and reactor scram occurred due to a high RV water level. The high RV water level was due to troubleshooting valve FV-642B using a general troubleshooting procedure. Since the procedure did not require a detailed work plan, the valve's air lockout was not reset prior to moving the valve's actuator handwheel. Contributing causes included a mis-orientated diaphragm pressure (actuator dome) gauge and an air leak in the air lockout device and actuator diaphragm. Corrective actions taken included disassembly, inspection and replacement of the FV-642B actuator.

For LER 89-028-00, the HPCIS was declared inoperable due to an overspeed trip. The overspeed trip was due to the failure of the ramp generator signal converter (RGSC) module that is part of the turbine speed control system. The RGSC was replaced and the turbine speed control system was calibrated.

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## ENERGY INDUSTRY IDENTIFICATION SYSTEM (EIIS) CODES

The EIIS Codes for this report are as follows:

### COMPONENTS CODES

Fuse FU

Switch, Pressure Ps

Regulating Device, Air Booster Relay 90

Valve, Injection (1301-50) INV  
Valve, Isolation (MO-1001-50) ISV  
Valve, Control, Flow (FV-642A/B, FV-643) FCV  
Control Operator, Flow, (HPCIS) FCO  
Power Supply, Electrical (640-42) JX  
Relay (16A-K29) RLY  
AirLock (FV-642A, FV-643) AL  
Device, Overspeed (RCICS Linkage) 12

## SYSTEMS CODES

Feedwater System SJ  
RCICS BN  
HPCIS BJ  
RWCU System CE  
Standby Gas Treatment System (SGTS) BH  
Feedwater Level Control System JB  
ESF Actuation System (PCIS/RBIS/RPS) JE  
Containment Isolation Control System (PCIS/RBIS) JM  
Reactor Building Environmental Control System (RBIS) VA  
Residual Heat Removal System (SDC Mode) BO  
Plant Protection System (RPS) JC

ATTACHMENT 1 TO 9010160047 PAGE 1 OF 1

10 CFR 50.73

BOSTON EDISON  
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October 2, 1990

BEC Co Ltr. 90-118

U. S. Nuclear Regulatory Commission

Attn: Document Control Desk

Washington, D. C. 20555

Docket No. 50-293

License No. DPR-35

Dear Sir:

The enclosed Licensee Event Report (LER) 90-013-00, "Manual Reactor Scram Due to Lockup of the Feedwater Regulating Valves", is submitted in accordance with 10 CFR Part 50.73.

Please do not hesitate to contact me if there are any questions regarding this report.

R. G. Bird

TM/bal

Enclosure: LER 90-013-00

cc: Mr. Thomas T. Martin  
Regional Administrator, Region I  
U. S. Nuclear Regulatory Commission  
475 Allendale Rd.  
King of Prussia, PA 19406

Sr. NRC Resident Inspector - Pilgrim Station

Standard BECo LER Distribution

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